

# Comparison of sodium and lead-cooled fast reactors regarding reactor physics aspects, severe safety and economical issues

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## Abstract

A large number of new fast reactors may be needed earlier than foreseen in the Generation IV plans. According to the median forecast of the Special Report on Emission Scenarios commissioned by the Intergovernmental Panel on Climate Control nuclear power will increase by a factor of four by 2050. The drivers for this expected boost are the increasing energy demand in developing countries, energy security, but also climate concerns. However, staying with a once-through cycle will lead to both a substantially increased amount of high-level nuclear waste and an upward pressure on the price of uranium and even concerns about its availability in the coming decades. Therefore, it appears wise to accelerate the development of fast reactors and efficient re-processing technologies.

In this paper, two fast reactor systems are discussed—the sodium-cooled fast reactor, which has already been built and can be further improved, and the lead-cooled fast reactor that could be developed relatively soon. An accelerated development of the latter is possible due to the sizeable experience on lead/bismuth eutectic coolant in Russian Alpha-class submarine reactors and the research efforts on accelerator-driven systems in the EU and other countries.

First, comparative calculations on critical masses, fissile enrichments and burn-up swings of mid-sized SFRs and LFRs (600 MW<sub>e</sub>) are presented. Monte Carlo transport and burn-up codes were used in the analyses. Moreover, Doppler and coolant temperature and axial fuel expansion reactivity coefficients were also evaluated with MCNP and subsequently used in the European Accident Code-2 to calculate reactivity transients and unprotected Loss-of-Flow (ULOF) and Loss-of-Heat Sink (ULOHS) accidents. Further, ULOFs as well as decay heat removal (protected Total Loss-of-Power, TLOP) were calculated with the STAR-CD CFD code for both systems.

We show that LFRs and SFRs can be used both as burners and as self-breeders, homogeneously incinerating minor actinides. The tight pin lattice SFRs (P/D = 1.2) appears to have a better neutron economy than wide channel LFRs (P/D = 1.6), resulting in larger BOL actinide inventories and lower burn-up swings for LFRs. The reactivity burn-up swing of an LFR self-breeder employing BeO moderator pins could be limited to 1.3\$ in 1 year. For a 600 MW<sub>e</sub> LFR burner, LWR-to-burner support ratio was about two for (U, TRU)O<sub>2</sub>-fuelled system, while it increased to approximately 2.8 when (Th, TRU)O<sub>2</sub> fuel was employed. The corresponding figures for an SFR were somewhat lower. The calculations revealed that LFRs have an advantage over SFRs in coping with the investigated severe accident initiators (ULOF, ULOHS, TLOP). The reason is better natural circulation behavior of LFR systems and the much higher boiling temperature of lead. A ULOF accident in an LFR only leads to a 220 K coolant outlet temperature increase whereas for an SFR the coolant may boil. Regarding the economics, the LFR seems to have an advantage since it does not require an intermediate coolant circuit. However, it was also proposed to avoid an intermediate coolant circuit in an SFR by using a supercritical CO<sub>2</sub> Brayton cycle. But in an LFR, the reduced concern about air and water ingress may decrease its cost further.

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## 1. Introduction

As predicted by the Special Report on Emission Scenarios (SRES) commissioned by the Intergovernmental Panel on Climate Change (IPCC), the primary energy demand will increase

between 1.7 and 3.7-fold until 2050 (INPRO, 2003). A mix of today's energy production options, however, falls short of the goal to provide a long-term, sustainable source of energy without adverse environmental/climate effects. In most of the SRES reference scenarios (see Fig. 1), the share of nuclear power is forecast to increase considerably by 2050, with a median of more than four times. However, a much larger growth of nuclear power would be needed, up to 15 times, if emissions of carbon dioxide are to be stabilized and then decreased beyond 2050–2060. In

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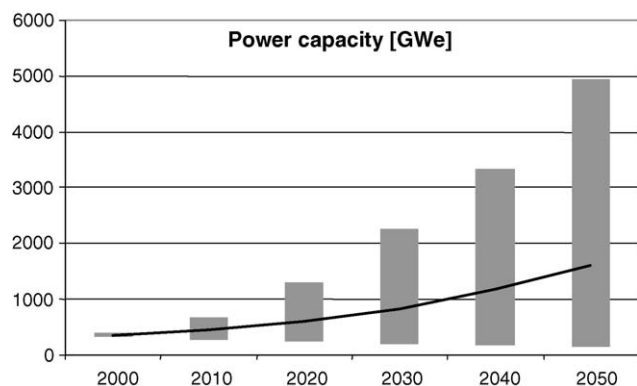


Fig. 1. Range of nuclear power capacities as predicted by 40 SRES scenarios. Solid line represents median (INPRO, 2003).

the long-term perspective, fusion power or other energy sources may be sufficiently developed, but probably too late to combat climate change.

New reactor designs, which will be commissioned as a replacement and expansion of the existing nuclear reactor parks, should be competitive, safe, proliferation resistant and meet the criteria of sustainability (GEN-IV, 2002). Next-generation systems also have to reduce the amount and radiotoxic inventory of residual wastes destined to geological repositories, hence addressing both public and scientific concerns regarding their long-term reliability. To achieve this goal, recycling of fuel and recovery of long-lived nuclides during reprocessing will be necessary. This will not only reduce the radiotoxic inventory of the waste but also allow converting long-lived actinides into energy.

An efficient use of fissile fuel resources together with the ability to burn its own high-level wastes and those coming from LWRs are primary design goals of new reactor designs developed under the auspices of the Generation IV initiative. Both self-breeder ( $CR \sim 1$ ) and ‘burner’ design options are therefore considered. The former system may operate in a pure fast reactor (FR) scheme while the latter will work in concert with LWRs or, in a double-strata scenario together with dedicated minor actinide (MA) burners (ADS). As the purpose of self-breeder and burner is different, their fuel composition will differ and hence also their neutronic and safety characteristics. An introduction of minor actinides (americium) into the fuel significantly deteriorates the reactivity coefficients (Doppler, coolant temperature/void reactivity).

In a self-breeder configuration, the uranium content will be adjusted such that the breeding gain is kept close to zero. Minor actinides will be recycled infinitely and their fraction in the equilibrium fuel should not exceed  $\sim 5$ – $10\%$  in order to keep the Doppler high and the coolant temperature reactivity coefficient low. In burners, which are assumed to destroy TRUs from a few (1–3) LWRs, the uranium content has to be limited in order to achieve high TRU consumption. An inert matrix is therefore applied instead of  $^{238}\text{U}$  to dilute highly reactive TRUs. Reactivity loss of self-breeder cores is small, allowing for extensive cycle lengths of several years. For example, in SVBR-75/100, the burn-up swing can be kept below 1\$ during 1 year (Toshinsky et al., 2002). The reactivity excess available in the compensation

rods is thus small excluding the possibility for a reactivity insertion leading to a prompt criticality. On the other hand, a large reactivity reserve is necessary in burner type cores (usually of several tenths of dollars).

The optimal choice of core materials for GEN-IV systems is still an open question. The requirement of a fast neutron spectrum for an efficient breeding and TRU incineration implies the usage of coolants with low moderating power, such as sodium, lead, lead/bismuth eutectic, or helium. In this study, two types of liquid metals proposed as coolants for GEN-IV systems are studied – lead and sodium.

## 2. Coolants

Basic properties of the considered coolants together with lead/bismuth eutectic are summarized in Table 1.

Sodium has superior thermal hydraulic properties, allowing for tight pin lattices. There is a large (but not always positive) experience with the operation of sodium-cooled fast reactors. While several power reactors have been shut down, BOR-60, JOYO, Phénix and BN-600 are still operating, the latter being in quasi-commercial operation since 1982. New sodium-cooled reactors are under construction in Russia, China and India.

Sodium features a reasonably low melting temperature, but also a low boiling point (1156 K), which raises safety concerns regarding unprotected transients leading to a coolant heat-up. Sodium exhibits high chemical activity with water, water vapor and air—a limited sodium leak and fire has stopped the operation of the Japanese MONJU reactor since 1995.

The choice of lead and lead-alloys as coolants is motivated on the one hand by their high boiling points, which avoids the risk of coolant boiling. On the other hand, lead and lead-alloys are compatible with air, steam,  $\text{CO}_2$  and water, and thus, no intermediate coolant loop is needed.

Lead/bismuth eutectic provides a low melting point (398 K), limiting problems with freezing in the system and features a low chemical activity with water and air excluding the possibility for fire or explosions. A drawback connected with lead/bismuth is the accumulated radioactivity (mainly due to the  $\alpha$ -emitter  $^{210}\text{Po}$ ,  $T_{1/2} = 138$  days), which could pose difficulties during fuel reloading or repair work on the primary circuit. However, IPPE Obninsk staff has developed methods to cope with the polonium during refueling and maintenance (Toshinsky, 2001).

Table 1  
Basic physical properties of liquid metal coolants

Coolant	Na	Pb	Pb/Bi
$\rho$ [ $\text{g}/\text{cm}^3$ ]	0.847	10.48	10.15
$T_m$ [K]	371	601	398
$T_b$ [K]	1156	2023	1943
$c_p$ [ $\text{kJ}/(\text{kg K})$ ]	1.3	0.15	0.15
$\rho c_p$ [ $\text{J}/(\text{m}^3 \text{K})$ ]	$1.1 \times 10^6$	$1.6 \times 10^6$	$1.5 \times 10^6$
$k$ [ $\text{W}/(\text{m K})$ ]	70	16	13
$v$ [ $\text{m/s}$ ]	10	2.5	2.5

Densities ( $\rho$ ), melting ( $T_m$ ) and boiling ( $T_b$ ) temperatures, specific heat ( $c_p$ ), thermal conductivities ( $k$ ) and maximum velocities ( $v$ ) are given at 700 K.

Lead is considered as a more attractive coolant option than lead/bismuth mainly due to its higher availability, lower price and lower amount of induced polonium activity (by a factor of  $10^4$ ), as given in a publication about the BREST-300 LFR reactor design (Adamov, 2001). Pure lead has a melting temperature of 601 K, which narrows in the reactor's operational interval to about 680–870 K. However, after more research, higher outlet temperatures might eventually be possible. Redundant electrical heaters are proposed in order to avoid problems with freezing and blockages in fresh cores.

Lead-alloy coolant velocities are limited by erosion concerns of protective oxide layers to about 2.5–3 m/s (Novikova et al., 1999). Typical sodium velocities are up to 8–10 m/s, hence lead has, in practice, a lower heat removal capacity, which requires higher pin pitch-to-diameter ratios to stay below cladding temperature limits. However, as shown later in this paper, these high pitch-to-diameter ratios enhance the natural circulation capability of the coolant, and thus, the safety performance of LFRs. On the other hand, SFR cores cannot have these large pitch-to-diameter ratios since their void worth would become too large (see later—e.g., Fig. 2).

Corrosion resistance of the structural material can be achieved through controlling oxygen content in lead or lead-alloy. This technology has been used in the Russian Alpha-class submarines and its effectiveness up to 820 K has been confirmed by the EU ADS research. The surface alloying by the so-called GESA method enhances corrosion resistance of the structural material further, at least up to 870 K (Wider et al., 2003). Higher temperatures could also be achieved by using ceramics instead of steels (e.g., SiC/SiC). It should also be noted that pure lead shows to be less corrosive than lead/bismuth eutectic at the same temperature (Wider et al., 2003).

Fast creep of a hanging reactor vessel during coolant heat-up transients is another important issue to be considered. It occurs significantly below the lead boiling point,  $\sim 1170$  K for SS-316, 1250 K for NIMONIC alloys and possibly higher for ODS steels. These values refer to an 11 m tall vessel.

Neutronically, the energy loss due to the elastic scattering in lead and lead/bismuth is significantly smaller than in sodium. However, due to the presence of several thresholds for inelastic scattering in lead-alloy coolants within the energy interval from 0.57 to 2 MeV, the neutron energy loss in inelastic scattering is notably larger than for sodium. Therefore, the neutron spectrum of lead and lead/bismuth-cooled reactors will be decreased for energies above 1 MeV (Tuček, 2004).

On the other hand, the magnitude of the neutron flux for a sodium-cooled reactor is decreased in the energy interval of 0.7–1.5 MeV, where contributions to the neutron slowing down from elastic and inelastic scattering reactions are nearly equal. Additionally, the neutron mean free path in sodium is larger than that of lead or lead/bismuth. Therefore, the leakage of neutrons and its contribution to the overall neutron balance in the system is more significant for sodium.

Further, higher scattering in lead and lead/bismuth without significantly increasing the moderation for neutrons below 0.5 MeV prevents the neutrons from escaping from the internal parts of the lead-alloy-cooled cores and, at the same time, pro-

Table 2

Pu/(U + Pu) fraction as a function of pitch-to-diameter ratio in a model critical 1200 MW<sub>e</sub> FR employing (U, Pu)O<sub>2</sub> fuel and cooled by sodium and lead/bismuth (Tuček, 2004)

P/D	Na	Pb/Bi
1.2	17.0	16.0
1.6	22.0	19.0
2.0	27.5	22.5
2.4	33.0	26.0

vide an excellent reflecting capability for the neutrons, which leak away from the core.

Hence, we can also infer that the neutron economy of the lead-alloy-cooled systems would be better than for sodium-cooled counterparts having the same geometry. For example, lead-alloy-cooled, (U, Pu)O<sub>2</sub>-fuelled systems require smaller plutonium enrichments than sodium counterparts to reach criticality, see Table 2.

### 3. Method for neutronic and burn-up calculations

The goal of this study is an indicative comparison of SFR and LFR cores that accommodate large fractions of minor actinides in the fuel and that feature similar safety coefficients (Doppler, coolant temperature reactivity). The former then ensures high MA consumption rates.

The Monte Carlo code MCNP was used for the calculation of the criticality, spatial distributions of neutron fluxes and power (Briesmeister, 2000). Reactivity coefficients were evaluated by using the perturbation model implemented in MCNP. Doppler reactivity feedback was estimated by evaluating a reactivity change upon the increase of fuel temperature from 300 to 1500 K. The MCB code was used to calculate fuel burn-up (Cetnar et al., 1998). Nuclear data libraries were adjusted for the temperature dependence using the NJOY code. The averaged temperatures of the core components were assumed as follows: 1500 K for fuel, 900 K for cladding and 600 K for coolant. The composition of the actinide vector is that of spent LWR UOX fuel—see Table 3. The fuel has a burn-up of 41 GWd/tHM and it is assumed to have undergone 30 years of cooling. Correspondingly, Pu/Np/Am fraction is then equal to 83/5/12. Depleted uranium (0.3% <sup>235</sup>U) is used in the analyses.

Table 3

Plutonium and minor actinide vector corresponding to the LWR UOX spent nuclear fuel with burn-up 41 GWd/tHM after 30 years of cooling

Isotope	Fraction
<sup>235</sup> U	0.003
<sup>238</sup> U	0.997
<sup>237</sup> Np	1.000
<sup>238</sup> Pu	0.023
<sup>239</sup> Pu	0.599
<sup>240</sup> Pu	0.264
<sup>241</sup> Pu	0.040
<sup>242</sup> Pu	0.074
<sup>241</sup> Am	0.871
<sup>243</sup> Am	0.129

In order to reach reasonable calculation times in MCB, we have chosen to adjust the system parameters (fissile enrichment) such that  $k_{\text{eff}}$  is one at BOC rather than at EOC. Our calculations thus somewhat underestimate the reactivity burn-up swing since the U/TRU fraction would have to be decreased in the latter case.

To give an indicative inter-comparison of both cores, a fuel cycle length of 330 days with 35 days refueling period was tentatively chosen.

#### 4. LFR and SFR design models

Both reactors have a power of 600 MW<sub>e</sub>, but a case of an LFR burner having an up-rated power level of 900 MW<sub>e</sub> was also investigated. For the lead-cooled fast reactor (LFR), the thermal efficiency is assumed to be 42% corresponding to an improved supercritical steam cycle (Cinotti, 2004). Similarly, a supercritical Brayton CO<sub>2</sub> cycle is being considered for sodium-cooled fast reactors (SFR), increasing the thermal efficiency to 45% (Schulenberg et al., 2003). However, as shown in recent studies, carbonates form when CO<sub>2</sub> reacts with sodium, which may cause some clogging of the primary circuit (Weaver, 2005).

The pellet and pin for the LFR concept cores differ for burner and self-breeder designs (Table 4). The active pin height is determined from a requirement to assure the thermo-mechanical stability of the pin column (limited bending) and achieve reasonable fuel burn-up rates/breeding. For the LFR self-breeder, an active pin length of 200 cm was used, similar to that proposed for the STAR-LM LFR design (ANL, 2005). For the LFR burner, a pancake-like, high-leakage core concept was chosen and the active pin length was only 100 cm. Pin and pellet diameter of LFR burner was the same as for BREST reactor (Adamov, 2001), while for LFR self-breeder these dimensions were larger as required by pin stability constraints in the heavy metal coolant environment. In order to reduce peak fuel temperatures in the LFR self-breeder, a 3-mm diameter concentric hole was used. Such central holes are used in all Russian LWRs and were also proposed for CAPRA and EFR. Pitch-to-diameter ratios of 1.5

(LFR burner) and 1.6 (LFR self-breeder) were used, which leads to low-pressure drop cores, enhancing natural circulation and hence increasing margins to core/vessel damage in ULOF and ULOHS accidents. In order to keep the axial temperature gradient in the coolant below 80 K (inlet and outlet temperatures 673 and 753 K, respectively), maximum coolant flow velocities of 2 m/s are necessary. This is well below the design limit of 2.5–3 m/s hence ensuring stability of the protective oxide layers. The height of the LFR vessel is kept at 11 m in order to ensure seismic stability of the reactor. The 600 MW<sub>e</sub> power, the 80 K coolant temperature gradient and the 11 m vessel height are based on the accepted European Lead-Cooled Fast Reactor System (ELSY) project (Cinotti, 2004).

The design considerations regarding SFR are based on the model of the WAC benchmark reactor (Wider et al., 1989). The axial and radial reflectors were removed and the active pin height is only 100 cm. As discussed above, sodium allows for higher coolant velocities than heavy metal coolants permitting tighter pin lattices of P/D ~ 1.2 for the SFR.

Homogeneous mixing of MAs into the fuel significantly deteriorates the coolant temperature and Doppler reactivity coefficients. A means to improve the coefficients is the tailoring of neutron spectra by moderators. In this study, we tentatively included BeO and TRIGA-type UZrH<sub>1.6</sub> pins in the fuel sub-assemblies. Hydrides have superior neutronic characteristics and efficiently thermalize the neutrons into the region of pronounced resonances, which significantly improve the Doppler. Nevertheless, their thermal properties are less favorable than those for BeO as, e.g., UZrH<sub>1.6</sub> seems to dissociate above 1070 K. On the other hand, besides its lower thermalizing capability, BeO is also highly toxic. Another option would be to use enriched <sup>11</sup>B<sub>4</sub>C as it was envisioned for CAPRA cores.

#### 5. LFR and SFR self-breeders

Self-breeder reactors feature low burn-up reactivity swing. Ideally, this should be below 1\$ over an extended time. Then, the consequences of reactivity-induced accidents are limited as

Table 4  
Design parameters of SFR and LFR core concepts considered in this study

Parameter	LFR burner	LFR burner	LFR breeder	SFR burner and breeder
Power (MW <sub>e</sub> )	600	900	600	600
Pellet outer radius (mm)	3.3		5.0	3.0
Clad inner radius (mm)	3.4		5.1	3.1
Clad outer radius (mm)	4.55		6.25	3.45
Pellet hole radius (mm)	–		1.5	–
Pitch-to-diameter ratio (P/D)	1.5		1.6	1.2
S/A outer flat-to-flat (cm)	20.10		23.60	14.66
Pins per S/A	217		127	271
Length of upper plenum (cm)	100		100	100
Length of lower plenum (cm)	10		10	10
Active pin length (cm)	100		200	100
Number of S/A	625		263	217
Averaged linear power (kW/m)	11.5	17.3	21.3	24.3

For LFR and SFR self-breeders, number of SAs and linear power correspond to unmoderated cores. In the case of LFR burner, linear power refers to BeO moderated core.

Table 5

Neutronic and burn-up characteristics of self-breeder LFR and SFR core designs fuelled by (U, TRU)O<sub>2</sub>

Reactor	Moderator	Number of moderator pins per SA	Number of core radial channels	Actinide mass at BOL (tHM)	Averaged TRU enrichment (%)	Doppler, $\Delta k$ (pcm)	Coolant, $\Delta k$ (pcm)	Burn-up swing $\Delta k$ per year (\$)
LFR	UZrH <sub>1.6</sub>	9	9	35.05	27.1	−92	57	−2.8
	BeO	18	11	48.15	27.5	−72	66	−1.3
	Unmoderated	–	9	37.62	24.1	−44	77	0.1
SFR	UZrH <sub>1.6</sub>	19	8	12.19	24.7	−95	52	−7.9
	BeO	45	10	16.39	26.1	−75	56	−3.5
	Unmoderated	–	8	13.09	22.9	−44	69	−2.6

Doppler and coolant temperature reactivity feedbacks correspond to the increase of fuel and coolant temperatures by 100 K. The burn-up swing corresponds to the first year of the start-up mode.

only a small amount of reactivity could be accidentally inserted into the reactor.

The SFR and LFR cores were divided into two and three enrichment zones, respectively. Uranium/TRU ratio in the individual zones was adjusted in order to attain criticality, while at the same time keeping radial power peaking factor below 1.3 at BOL. The duct-free sub-assembly structure was modeled explicitly in MCNP.

### 5.1. Neutronic and burn-up performance

As seen in Table 5, where neutronic and burn-up characteristics of self-breeder cores are summarized, the cores with UZrH<sub>1.6</sub> moderator have Doppler reactivity feedback about 60–80% higher than coolant temperature reactivity coefficient. The reactivity feedback for a BeO moderated core is worse, although the Doppler still remains larger than the coolant temperature reactivity feedback for both LFR and SFR. The unmoderated cores have significantly deteriorated reactivity feedbacks, but feature favorable burn-up characteristics (very low reactivity swing especially in the case of LFR). Hydride materials are hence the most effective in tailoring the neutron spectra—increasing Doppler by moderating neutrons into the region of large actinide resonances and decreasing coolant temperature reactivity feedback by diminishing the spectral gradient during coolant heat-up/voiding.

Both LFRs and SFRs accommodate about the same amount of MAs in the fuel (~5%) at comparable reactivity coefficients (Table 5), but the burn-up reactivity swing of SFR cores is larger. This is due to a lower breeding ratio and also the lower actinide inventories of SFR core designs. However, this also means that the actinide burn-up rate is larger in an SFR than an LFR. For LFR self-breeders, indications are that MA consumption is significant and amounts to about 70 kg/year.

As shown by Feldman et al. (2004), low power density LFRs can have very long refueling intervals of up to 15 years, which is ideal for remote locations and developing countries without nuclear infrastructure. Modular and easily transportable LFRs would be best suited for these applications.

#### 5.1.1. Optimizing the lattice pitch-to-diameter ratio

One of the attractive options aiming to improve safety of SFRs is to enhance coolant natural circulation behavior. This can be

achieved by minimizing the pressure drop, e.g., by enlarging the pin lattice and using a simple flow path design. However, when increasing P/D, the enlarged fission probability of even neutron numbered actinides (due to larger spectral hardening) exacerbates sodium coolant void worth. The impact of enlarged P/D ratios on sodium void worth was therefore investigated (see Fig. 2).

By applying in-core moderators, coolant void worth can be kept reasonably low even for cores with slightly enlarged pin lattices. For P/D of 1.4, the core void worth remains below 5.5\$ for an SFR using UZrH<sub>1.6</sub> moderator while it is about 6.5\$ for an SFR with BeO. Enlarging P/D even further lead to a significant increase of the coolant void; for P/D=1.8 it is about 7\$ for a UZrH<sub>1.6</sub> moderated core. Note that in burner type of cores void worth could be improved by spoiling of the core geometry (pancake-like core) as requirements with regard to the neutron economy are less stringent (no breeding).

#### 5.1.2. Neutronic and burn-up performance of Th-based fuels

Similarly to our previous studies, which considered U-based fuels, neutronic and burn-up performance of (Th, TRU)O<sub>2</sub> in

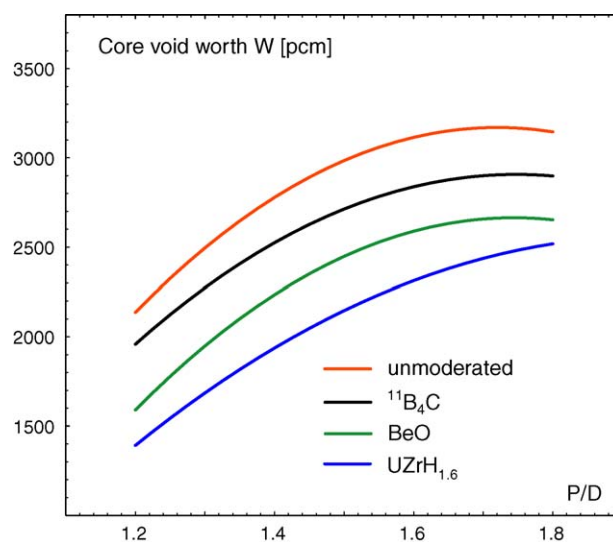


Fig. 2. Sodium void worth as a function of pitch-to-diameter ratio. In this study, CAPRA pellet and pin designs were used (Conti et al., 1995). Fuel-to-moderator volume ratio was ~5.

Table 6

Neutronic and burn-up characteristics of LFR and SFR core designs with (Th, TRU)O<sub>2</sub> fuel

Parameter	LFR	SFR
Average TRU fraction in the fuel (%)	30	29
Number of moderator pins per S/A	16	34
Number of core radial channels	9	8
$m_{\text{act}}$ at BOL (tHM)	36.29	12.27
Doppler, $\Delta k$ (pcm)	−113	−125
Coolant, $\Delta k$ (pcm)	45	40
Burn-up swing (\$/year)	−0.4	−5.6

Doppler and coolant temperature reactivity feedbacks correspond to the increase of fuel and coolant temperatures by 100 K. The burn-up swing corresponds to the first year of the start-up mode.

SFR and LFR was investigated. In this case, ThZrH<sub>1.6</sub> moderator was also used instead of UZrH<sub>1.6</sub>. Indeed, thorium-fuelled systems feature low self-production of plutonium and minor actinides, improving TRU transmutation rates. However, the reprocessing of Th-based fuels seems to be more complex than that of (U, TRU)O<sub>2</sub>. In aqueous technologies, thorium extraction is poor and a salting-out agent is required. Moreover, little is known about performance of pyro-chemical reprocessing (e.g., fluoride volatility method) and a considerable R&D program would have to be undertaken.

LFRs feature very low burn-up reactivity swing, less than 1\$ per year, due to breeding <sup>233</sup>U (Table 6). Annually, 348 kg of plutonium and 96 kg of minor actinides are transmuted and 246 kg of <sup>233</sup>U produced. These figures correspond to the first cycle of the start-up core. For an equal power SFR, the corresponding transmutation rates are 322 kg/year of Pu and 95 kg/year of MA and at the same time as 210 kg/year of <sup>233</sup>U are produced.

## 5.2. Safety performance

Safety analyses were performed with the European Accident Code-2, EAC-2 (Wider, 1990) and the STAR-CD (STAR-CD, 2004) code. The accidents considered were unprotected Loss-of-Flow (ULOF) and Loss-of-Heat-Sink (ULOHS) as well as protected Total Loss-of-Power (TLOP) accidents.

### 5.2.1. Unprotected Loss-of-Flow accidents

The main parameters of the SFR under consideration are described in Table 4. In this ULOF calculation, the WAC benchmark reactor was downgraded from 800 to 600 MW<sub>e</sub> (or 1426 MW<sub>th</sub>) for reasons of comparison with an LFR of the same power. The flow coast-down is described by the equation

$$\frac{G(t)}{G_0} = \frac{1}{1 + t/t_c} \quad (1)$$

where  $G(t)$  and  $G_0$  are coolant flow rates at time  $t$  and  $t = 0$ , respectively,  $t_c$  is equal to 6 s.

As the Doppler and axial fuel expansion cannot compensate for positive reactivity inserted during coolant heat-up, the sodium starts boiling after 22.3 s (see Fig. 3).

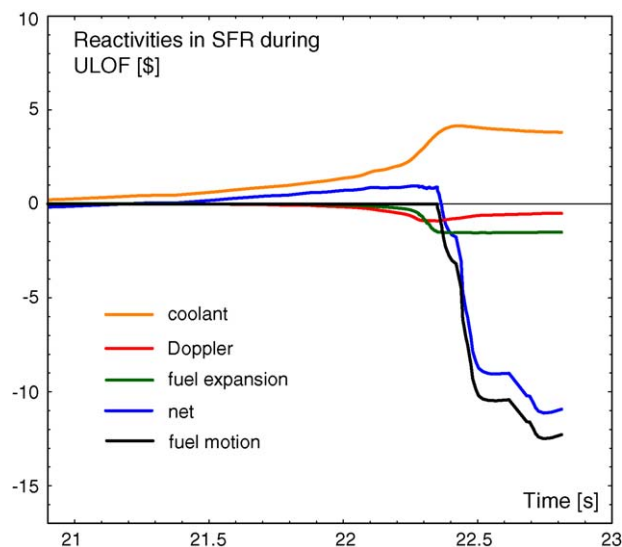


Fig. 3. Reactivities in SFR-ULOF accident.

At this stage, the flow is still 20% of nominal. It has been shown in earlier studies that in a smaller SFR design (800 MW<sub>th</sub> or about 335 MW<sub>e</sub>), a sizable negative radial expansion feedback of the structure and a reasonable natural circulation are needed to prevent boiling (Van Tuyle et al., 1989). For the Super-Phénix reactor an increase in grace period of about 100 s could be achieved using a constantly rotating flywheel that provided temporary inertia to the pump during a ULOF.

Fig. 3 shows that sodium boiling leads to a significant positive reactivity insertion increasing the power (Fig. 4), which leads to more boiling and voiding until fuel melts. Then, fuel pins breach and molten fuel is swept out shutting the excursion down after about half the core is molten. The fuel feedbacks could also be temporarily positive if pin failures occur near the mid-plane. The present calculations were performed with the

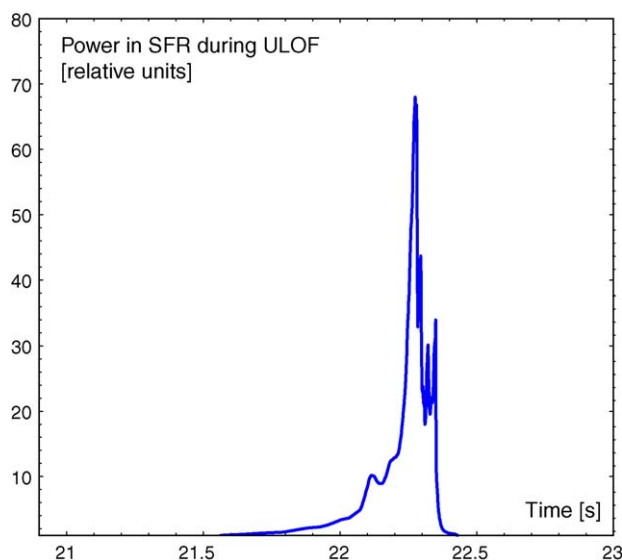


Fig. 4. Relative power in SFR-ULOF accident. The power peak is due to boiling of sodium and insertion of large positive reactivity as a consequence. The calculations were performed with the EAC-2.

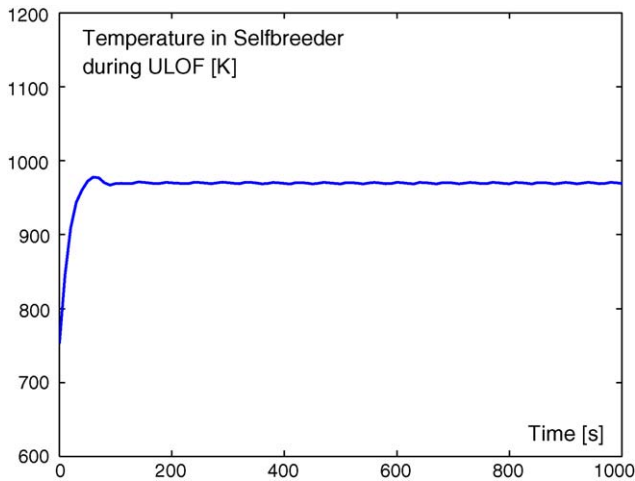


Fig. 5. Above-core averaged coolant temperatures in LFR-ULOF accident. The core coolant outlet temperature peaks at 970 K. The calculation was performed with STAR-CD.

European Accident Code-2 using 10 radial calculational channels. There are new claims that relatively fast radial structural feedbacks can prevent boiling and power peaks as shown in Fig. 4. The IAEA-INPRO is interested in starting a research project on this important safety issue. Another possibility to reduce the sodium voiding reactivity is the exclusion of axial blankets.

Fig. 5 concerns the behavior of an LFR of 600 MW<sub>e</sub> in a ULOF accident. The same flow coast-down as in the previous SFR calculation was applied, see Eq. (1). The inlet temperature was kept constant, as only the core and primary circuit were modeled. No negative feedbacks are considered since the STAR-CD CFD code does not include neutronic models. It was found that the average coolant outlet temperature peaks at 970 K, i.e., the outlet temperature increases by about 220 K. This poses no problems regarding coolant boiling and, in short-term perspective, no corrosion problems will occur either. The natural convection flow rate is 22% of nominal.

A corresponding EAC-2 calculation considering feedbacks showed that the power drops within 15 s to 80% of nominal power and stays there. The small negative Doppler and axial fuel expansion feedbacks offset the small positive coolant reactivity insertion.

This remarkable behavior of the LFR is due to the low-pressure drop core ( $P/D = 1.6$ ) and a simple flow path design as in the Ansaldo ADS (Cinotti et al., 2001) and the ANL STAR-LM designs (ANL, 2005). In all these designs, the coolant rises above the core and then continues down through a down-comer (where the heat exchangers are located) and back to the core. In the ANL design, solely natural circulation cooling is used even for regular, steady state operation.

In Fig. 6 it is shown how SFRs would behave during LOFs in simple flow-path designs and without negative feedbacks. For a  $P/D$  of 1.2, the averaged coolant outlet temperature would exceed the boiling point in about 20 s (see also Fig. 3). The main reason why the SFR did not behave similarly as the LFR is the higher pressure drop of the SFR core. However, by increasing  $P/D$  to

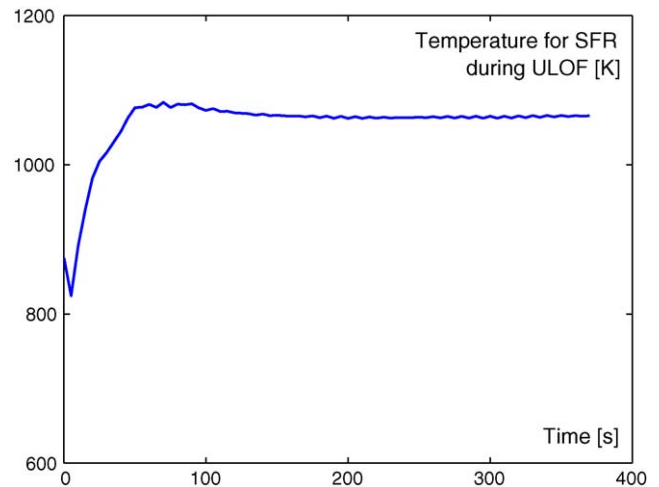


Fig. 6. Averaged outlet coolant temperatures in SFR-ULOF with large  $P/D = 1.8$ .

1.8, the SFR shows essentially the same temperature behavior as the LFR (Fig. 5) during a ULOF. The only difference is the 110 K higher initial outlet temperature of the SFR under consideration. But this high  $P/D$  leads to a high void worth (see Fig. 2).

#### 5.2.2. Unprotected Loss-of-Heat Sink accidents

In this type of accident, the heat removal by the secondary circuit ceases and the core heats up. An EAC-2 calculation was performed for which the inlet temperature increase was first calculated by STAR-CD for full power and then modified according to the changing reactor power as calculated by EAC-2.

As can be observed in Figs. 7 and 8, the power slowly drops to about 15% of full power during 1000 s (about 16 min). The coolant and cladding temperatures rise to about 1100 K. In this type of accident, the grid plate expansion will give an additional negative reactivity feedback bringing the power down to a level where the emergency decay heat removal can remove the residual heat. Note that the grid plate expansion is not yet taken into account in EAC-2.

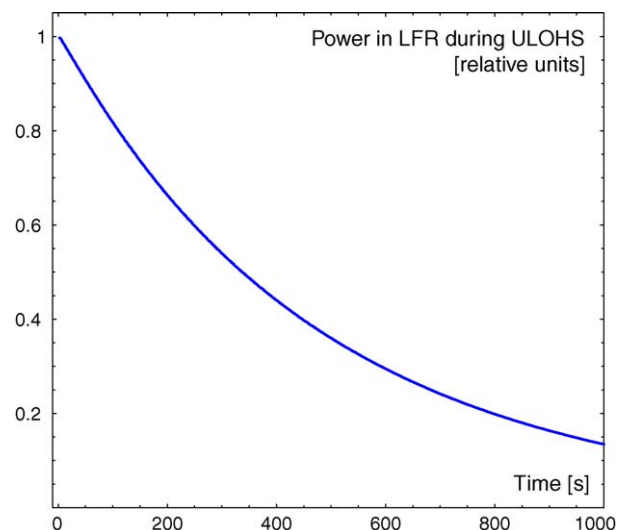


Fig. 7. Power relative to the nominal during ULOHS in LFR self-breeder.

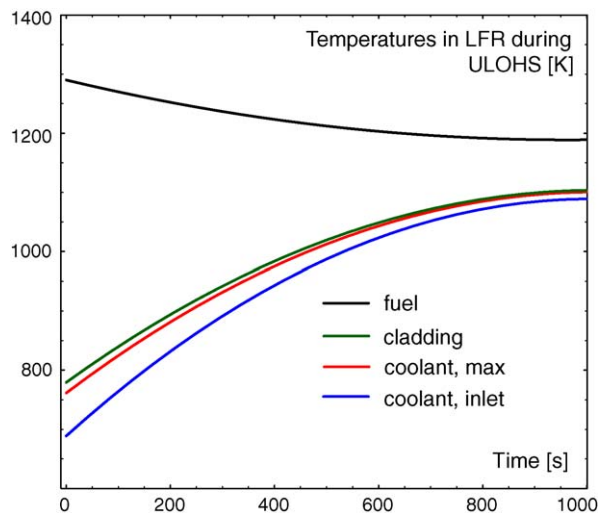


Fig. 8. Temperatures of fuel, cladding and coolant in LFR self-breeder during ULOHS.

The same accident simulation in an SFR gives a 40% faster temperature increase (also if grid plate expansion is disregarded). If the corresponding reactivity effect is not strong enough, the SFR will reach the coolant's boiling point of 1156 K.

### 5.2.3. Total Loss-of-Power accidents

In a protected TLOP accident where diesel-driven generators are unavailable, emergency decay heat removal becomes important. A Reactor Vessel Air Cooling System (RVACS) for the two 600 MW<sub>e</sub> systems was investigated (Carlsson, 2000).

Fig. 9 shows that the LFR self-breeder reaches within 90 K of the steel creep limit after 2 days. Thus, a more efficient emergency cooling such as an IRACS (In Vessel Reactor Auxiliary Cooling System) may be needed. Also a water pool surrounding the guard vessel, as has been proposed by Toshinsky et al. (2002), is an interesting option for the emergency cooling. For the 600 MW<sub>e</sub> SFR under consideration, the TLOP eventually led

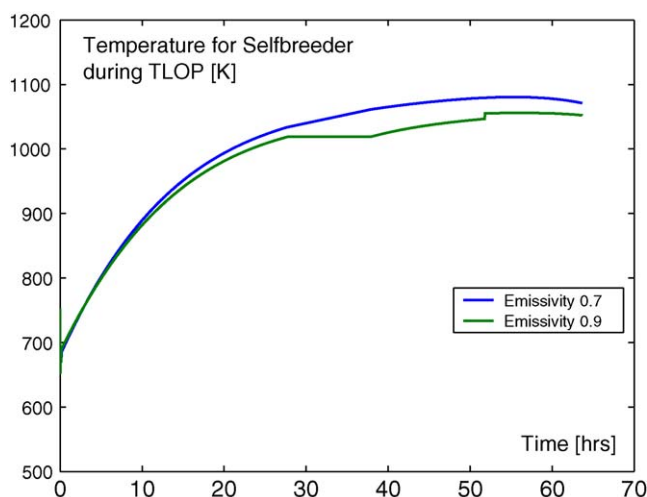


Fig. 9. Vessel temperature evolution in LFR self-breeder during TLOP, heat removal is provided by RVACS. The calculations were performed with STAR-CD.

to sodium boiling. Again, RVACS should not be used for such a large power reactor—IRACS would also be needed.

### 5.2.4. Reactivity accidents

Regarding reactivity insertion accidents the LFR has an advantage due to lower burn-up swings (see Tables 5 and 6). In addition, in the case of large reactivity insertions that lead to pin failures, the fuel sweep-out is more likely in LFRs due to a low pressurization during fuel-coolant interaction and due to the much larger inertia of lead. This is supported by the findings during an inlet blockage accident in the first Russian submarine driven by lead/bismuth-cooled reactors. When the reactor power became low, probably due to negative feedbacks, it was attempted to get the power up again by withdrawing the control rods. When a control rod was withdrawn, the power went up and then down again to a low value. This was repeated several times until all control rods were withdrawn. When the reactor vessel was opened several years later, it was found that fuel fragments were dispersed in lead/bismuth (Gromov et al., 1998).

## 6. LFR and SFR burners

In order to maximize the burn-up rate of transuranics from LWRs and limit breeding, the uranium fraction in the reactor core has to be significantly reduced in comparison to standard FBR designs. The excess reactivity of highly reactive TRU fuel has to be subsequently compensated by an introduction of a diluent (inert matrix) or a neutron absorber.

In our previous study (Wider et al., 2005), it was shown that an MgO matrix features favorable neutronic characteristics, limiting the amount of reactivity introduced into the system during coolant heat-up. However, recent experiments have shown that MgO-based TRU fuels have stability problems above 2140 K (Haas, 2005). Therefore, in this paper, an innovative CERMET (U, TRU)O<sub>2</sub>-<sup>92</sup>Mo fuel was chosen as a fuel candidate for SFR and LFR burners. Molybdenum is enriched in <sup>92</sup>Mo in order to lower parasitic neutron absorption and reduce production of long-lived <sup>99</sup>Tc.

### 6.1. Neutronic and burn-up performance

The volume fraction of <sup>92</sup>Mo in (U,TRU)O<sub>2</sub>-<sup>92</sup>Mo fuel was kept constant at 50% ensuring fabricability and thermal stabil-

Table 7  
Neutronic and burn-up performance of SFR and LFR burners

Parameter	LFR		SFR
Power (MW <sub>e</sub> )	600	900	600
Average TRU fraction in the fuel (%)	50		43
Number of moderator pins per S/A	19		19
Number of core radial channels	14		9
$m_{act}$ at BOL (tHM)	17.07		7.56
Doppler, $\Delta k$ (pcm)	-50		-54
Coolant, $\Delta k$ (pcm)	38		36
Burn-up swing (\$/year)	-11.7	-16.9	-23.2

Doppler and coolant temperature reactivity feedback for LFR and SFR burner cores corresponding to the increase of fuel and coolant temperatures by 100 K. The burn-up swing corresponds to the first year of the start-up mode.

Table 8

Economic comparisons of LFR, SFR and gas-fired plant (Zrodnikov et al., 2003)

Energy system considered	SVBR-75/100 LFR	BN-800 SFR	Gas PGU-325
Number of plants $\times$ power (MW <sub>e</sub> )	16 $\times$ 102	2 $\times$ 890	5 $\times$ 325
Efficiency of the net plant (%)	34.6 <sup>a</sup>	46.2	44.4
Specific capital investments (\$/kW, price of 1991)	661.5	783.4	600
Cost of electricity (cent/kWh, price of 1991)	1.46	1.56	1.75

<sup>a</sup> Will be higher for supercritical steam cycle.

ity of the fuel. The core designs thus resemble a characteristic configuration of a TRU-burner operating together with LWRs in a two-component scheme. BeO moderator was applied in both cases.

In order to achieve criticality, the average TRU fraction in the LFR core has to be *actually* slightly higher than for the SFR despite more than twice the actinide mass present in the LFR (Table 7). The reason is the tight pin lattice of SFR ( $P/D = 1.2$ ), which offers better neutron economy than the wide channel LFR design. The burn-up swing of the burner cores was roughly proportional to core power and inversely proportional to the initial actinide mass, which means that for a given reactivity excess in the shim rods at BOL the frequency of outages in an LFR is about half compared to an SFR. A slight deviation from the proportionality is due to different fractions of TRUs in the fuel resulting in different breeding.

Somewhat stronger Doppler reactivity feedback is provided in the SFR burner than in lead-cooled cores in which less neutrons are scattered down to the region of pronounced resonances and where the <sup>238</sup>U fraction is lower. SFR burners (600 MW<sub>e</sub>) can annually transmute 263 kg of plutonium and MAs, which roughly corresponds to an annual production of transuranics of a 1 GW<sub>e</sub> LWR with a fuel burn-up of 41 GWd/tHM. For a 600 MW<sub>e</sub> LFR, the corresponding figure is 303 kg. The initial charge of the 600 MW<sub>e</sub> LFR and SFR is about 8 and 3.3 core loads of TRUs from 1 GW<sub>e</sub> LWRs, respectively. This means that one can “park” considerably more “waste” in an LFR. For a 900 MW<sub>e</sub> LFR burner, the amount of TRUs transmuted annually is understandably also higher and equal to 449 kg.

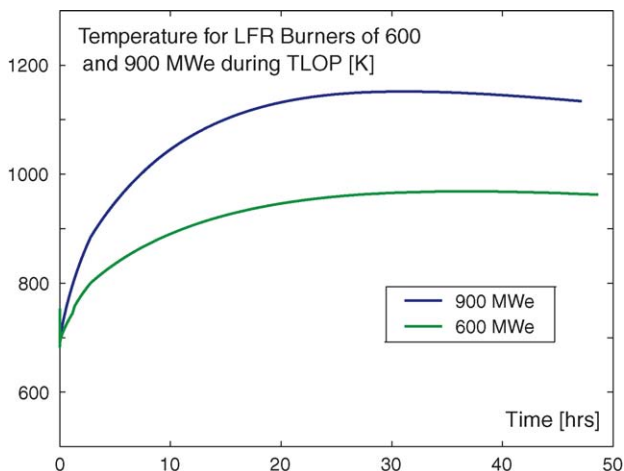


Fig. 10. Vessel temperature evolution in LFR burners during TLOP, heat removal is provided by RVACS. The calculations were performed with STAR-CD.

## 6.2. Safety performance

Thanks to the flat (pancake-like) geometry, low power density and the low-pressure drop of the LFR burner, the coolant temperature increase during ULOF accidents is mild. Without any reactivity feedbacks taken into account the 900 MW<sub>e</sub> reactor stabilizes at about 880 K, compared to 830 K for the 600 MW<sub>e</sub> core.

For the same 900 MW<sub>e</sub> core during the TLOP accident, the temperature is very close to the 316-steel fast creep limit and reaches 1150 K when a regular reactor vessel air-cooling system (RVACS) is employed (see Fig. 10). However, note that more efficient passive decay heat removal can be achieved by, e.g., an IRACS.

## 7. Economic aspects of SFR and LFR

Regarding economics, fast reactors were earlier considered more expensive to build and their electricity generation cost higher than that of LWRs. However, in the last few years several Russian publications have indicated that the lead/bismuth-cooled SVBR-75/100 is cheaper to build than any other reactor type and that the electricity generation cost is even lower than that of gas-fired plants (Table 8).

The reasons for this are that no intermediate coolant loop is needed for an LFR and that less safety-related systems have to be included thanks to its inherent safety. The prolonged, 8-year fuel cycle increases availability and helps to reduce the electricity generation cost, too. Note that in a true LFR, lead would be used instead of lead/bismuth. Since lead is about 10 times cheaper than lead/bismuth, the capital cost for an LFR may be even lower than for the SVBR-75/100. Yet another aspect would make an LFR more economical—there is less concern about water or air ingress or leaks, the latter can cause a fire in an SFR. Also a larger LFR would probably be economical considering the economy of size.

If a supercritical CO<sub>2</sub> Brayton cycle can be used in an SFR, the intermediate cooling loop can be omitted and the SFR could also become more economical. However, as mentioned above, there is a question about orifice clogging by Na–CO<sub>2</sub> reaction products. As an alternative, N<sub>2</sub> gas is also considered for the secondary circuit instead of CO<sub>2</sub>.

## 8. Conclusions

Considering the typical core lattice design configurations, the LFR showed advantages over SFR regarding behavior in severe

accidents like ULOF, ULOHS and TLOP. This is due to the better natural circulation behavior of an LFR design and the much higher boiling temperature of lead. Moreover, the chemical inactivity of lead excludes the possibility of fires or other strongly exothermic reactions with air, water and water vapor. This means that an LFR is a very robust system regarding safety. A drawback of the LFR is that for present steels a protective oxide layer or a coating is needed to minimize erosion during normal operation. These layers and coatings are sensitive to coolant temperatures above 870 K for prolonged periods of time. For LFRs, corrosion and material characteristics of steels should be further investigated, in particular under irradiation conditions. Moreover, totally innovative materials that appear promising, as, e.g., SiC/SiC, are already being studied.

An LFR appears also to have an economic advantage since it does not need an intermediate coolant circuit and the number of reactor outages can be limited because of the smaller burn-up swing. The latter aspect is also relevant for the remote location of modular LFRs. However, SFRs could also become more economically attractive if an inert gas could be used as a secondary coolant. Moreover, there is considerably more experience with sodium than lead or lead-alloys although the experience with sodium was not always satisfying.

LFRs and SFRs can be used both as burners and as self-breeders; LFRs having some advantages over SFRs (e.g., having a higher TRU consumption). However, SFRs feature larger actinide burn-up rates than LFRs mainly due to lower actinide inventories. LFR burners (600 MW<sub>e</sub>) can incinerate transuranic wastes from about two LWRs of the same power output. A corresponding figure for an SFR is slightly lower and equals to 1.7. When Th-based fuels are employed, the LWR-to-burner support ratio increases to approximately 2.8.

However, the burning of plutonium will not be reasonable if a rapid expansion of nuclear power is considered important, e.g., because of climate change concerns, energy security and economic resource utilization. Moreover, as indicated in this paper, it appears that a substantial minor actinide consumption can also be achieved in self-breeders.

We showed that Doppler and coolant temperature reactivity coefficients could be improved by moderating pins placed heterogeneously in all fuel sub-assemblies. Owing to the chemical toxicity of beryllium, hydrides should be considered as an alternative moderator material that, additionally, provides better neutronic characteristics. In this respect we note that instead of UZrH<sub>1.6</sub>, CaH<sub>2</sub> seems to be the most promising candidate nowadays since it appears to be more stable and can operate at higher temperatures ( $T_{\max} \approx 1100$  K). In order to improve the stability of hydrides, one could consider, e.g., encapsulating of the pellets and/or coating of cladding internal surfaces by tungsten or molybdenum. These options are currently under investigation.

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